

ACCESSION #: 9603250463
LICENSEE EVENT REPORT (LER)

FACILITY NAME: WATTS BAR NUCLEAR PLANT - UNIT 1 PAGE: 1 OF 12

DOCKET NUMBER: 05000390

TITLE: Manual Reactor Trip and Related Engineered Safety Feature
Actuations

EVENT DATE: 02/19/96 LER #: 96-004-00 REPORT DATE: 03/19/96

OTHER FACILITIES INVOLVED: DOCKET NO: 05000

OPERATING MODE: 1 POWER LEVEL: 15

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR
SECTION:

50.73(a)(2)(iv)

LICENSEE CONTACT FOR THIS LER:

NAME: Rickey Stockton, Compliance TELEPHONE: (423) 365-1818
Licensing Engineer

COMPONENT FAILURE DESCRIPTION:

CAUSE: B SYSTEM: SD COMPONENT: LT MANUFACTURER: R369
B SD LR G080

REPORTABLE NPRDS: N
N

SUPPLEMENTAL REPORT EXPECTED: NO

ABSTRACT:

On February 19, 1996, at approximately 1502 EST, the operators in the Main Control Room (MCR) manually tripped the main turbine due to a loss of normal feedwater and a corresponding decrease in steam generator (SG) level. The motor driven auxiliary feedwater (MDAFW) pumps were started to recover SG levels. At approximately 1504 EST, the reactor was manually tripped when SG levels could not be recovered. During this event, the turbine driven auxiliary feedwater (TDAFW) pump started and aligned to essential raw cooling water. With prompt operator action, the SG levels were recovered and the plant stabilized in Mode 3.

The cause of this event was the failure of the hotwell level control system to maintain an adequate water supply for the hotwell pump's net positive suction head due to a design deficiency. The deficient design

involved the sense line attachment location to the condenser which provided a false indication and signals to control systems of high hotwell level as the main turbine is loaded. The automatic start of the TDAFW pump was as designed. However, the cause of the alignment to essential raw cooling water was determined to be pressure switch actuations due to the MDAFW pump pulsations experienced at low flow conditions.

Corrective actions include design and modifications to the hotwell level indication, operating procedure enhancements for maintaining hotwell level during reactor startup conditions, and modifications to the TDAFW pressure switch control logic.

END OF ABSTRACT

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I. PLANT CONDITIONS

At the time of the event, Watts Bar Unit 1 was in Mode 1 at normal operating temperature of 557 degrees F and pressure of 2235 psig. TVA operators were in the process of slowly loading the main generator (Energy Industry Identification System (EIIS) Code TG) to 15 percent load. Actual main generator output was approximately 95 MWe and reactor power was about 15 percent. Hotwell pumps 1A and 1B (EIIS Code SD/P), condensate booster pump 1A (EIIS Code SD/P), and the main feedwater pump 1A (EIIS Code SJ/P) were in service supplying feedwater to the steam generators (SGs) (EIIS Code SG).

II. DESCRIPTION OF EVENT

A. Event

At 1314 EST on February 19, 1996, TVA operators started rolling the main turbine (EIIS Code TRB) in anticipation of loading the generator. At 1410 EST, operators observed hotwell pump 1A suction strainer (EIIS Code STR) delta P decreased 2-3 inches of water. At 1418 EST, the main generator was synchronized and tied to the grid.

At 1432 EST, the hotwell level alarm (EIIS Code LA) sounded in the Main Control Room (MCR). Operators acknowledged this alarm and cleared it at 1434 EST. At 1436 EST, the hotwell hi/lo alarm (EIIS Code LA) annunciated in the MCR. Both the MCR readings and those taken from the local sight glass indicated high level.

At 1443 EST, automatic dump back from the hotwell to the condensate storage tank (CST) (EIIS Code KA/TK) increased significantly. At the same time, an Assistant Unit Operator (AUO) stationed locally at the hotwell pumps observed increasing delta P at strainers for hotwell pumps A and B. However, between 1450 and 1457 EST, this AUO witnessed the hotwell pump A1 delta P readings go to 0. In addition, he observed suction head not changing and discharge pressure at 250 psig and decreasing. At this time, the AUO notified the MCR. Concurrent with these local observations, the MCR unit operator (UO) started hotwell pump 1C (EIIS Code SD/P). Subsequently, the UO secured hotwell pump 1A. At 1457, the UO observed condensate booster pump (CBP) 1A had tripped due to low suction pressure. The UO also noticed that hotwell pump discharge pressure had increased momentarily then decreased when CBP tripped.

At 1458 EST, the steam generator low level alarms (EIIS Code SG/LA) sounded. At 1459 EST, the Assistant Shift Operations Supervisor (ASOS) directed turbine load reduced. Concurrently, the UO observed hotwell level high but returning to normal. At 1501 EST, CBP 1A restarted, but was manually stopped after 50 seconds when the operator observed the ammeter (EIIS Code II) for the motor pegged high. After decreasing the generator load from approximately 95 megawatts (MWe) to 60 MWe, the UO was directed to manually trip the main turbine. At 1503 EST, the

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UO manually started the motor driven auxiliary feedwater pumps 1A-A and 1B-B when steam generator levels were observed to be at approximately 20 percent.

At 1504 EST, when the steam generator levels were observed to be at approximately 17 percent, the UO manually tripped the reactor (EIIS Code RCT). At this point, main feedwater isolation occurred. In addition, motor driven auxiliary feedwater pump suction pressure switches (EIIS Code BA/PS) alarmed (EIIS Code PA) and the turbine driven auxiliary feedwater (TDAFW) pump (EIIS Code TRB/BA/P) started.

At 1506 EST, the LoLoT sub ave (EIIS Code LA/TA) alarm sounded. At 1508 EST, the UO adjusted MDAFW pump level control valves (EIIS Code MO/BA/P/LCV) to reduce AFW flow and to control the cooldown per procedure ES-0.1, "Reactor Trip Response."

At 1510 EST, a UO observed one set of essential raw cooling water (ERCW) valves (1-FCV-3-136A and 1-FCV-3-136B) (EIIIS Code BI/FCV) on the TDAFW had opened and had received multiple alarms (EIIIS Code PA) on AFW pressure switches (EIIIS Code PS) to TDAFW pump. At 1512 EST, after verifying adequate heat sink capability, the ASOS directed the UO to stop the TDAFW pump. The UO then closed the ERCW valves to the TDAFW pump.

At 1516 EST, the hotwell hi/lo level alarm (EIIIS Code LA) cleared. Between 1517 and 1522 EST, low level alarms on the steam generators cleared. During this time, the UOs closed the main steam isolation valves (MSIVs) (EIIIS Code SB/ISV) to reduce cool down. At 1714 EST, the MSIVs were reopened. At 1729 EST, the UOs started the standby main feedwater pump and subsequently secured the MDAFW pumps 1A-A and 1B-B (EIIIS Code MO/BA/P). With the plant stabilized in Mode 3, an incident investigation team was assembled to determine the cause of this event.

B. Inoperable Structures, Components, or Systems that Contributed to the Event

There were no inoperable structures, components, or systems that contributed to this event.

C. Dates and Approximate Times of Major Occurrences

The following events happened on February 19, 1996:

TIME EVENT

1314 Started rolling main turbine.

1410 Hotwell pump 1A suction strainer delta P decreased 2-3 inches of water.

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1418 Main generator synchronized and tied to grid.

1432 Hotwell level alarm.

1434 Hotwell level clear.

1436 Hotwell hi/lo alarm. (MCR indications and local sight

glass indicate hi level).

1443 Auto dump back from hotwell to CST increases significantly.

(Approximately)

1443 AUO at hotwell pumps observed increasing delta P at strainers for hotwell pumps A & B.

1450-1457 AUO observed delta P readings go to 0 on 1A hotwell pump and attempted to notify the MCR by radio, but had to go to a telephone because MCR could not understand the transmission. AUO also observed suction head not changed and discharge pressure at 250 and decreasing.

1456 UO started Hotwell pump 1C due to low hotwell pump discharge pressure.

1457 Secured Hotwell pump 1A.

1457 Operators observed CBP 1A trip on low suction pressure (Hotwell pump discharge pressure increased momentarily then decreased when CBP tripped).

1458 SGs level low, alarms started coming in.

1459 ASOS directed turbine load reduced.

1459 Operator observed hotwell level high but returning to normal.

1501 CBP 1A restarted and manually stopped after 50 seconds.

1501 Operators tripped turbine after reducing load from approximately 95 MW to 60 MW.

1503 Operators started MDAFW pumps 1A-A and 1B-B at approximately 20% SG level.

1504 Reactor was tripped manually. SG level was approximately 17% (narrow range). Auto main feedwater isolation occurred. All control rods fully inserted.

1504 AFW Pump pressure switches alarmed. Turbine driven AFW (TDAFW) started.

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1506 Lo Lo T sub ave alarmed.

1508 The MDAFW pumps level control valves are adjusted to control AFW flow and to control cool down per procedure ES-0.1.

1510 Operators observed one set of Essential Raw Cooling Water (ERCW) valves, 1-FCV-3-136A and 1-FCV-3-136B, on TDAFW had opened and multiple alarms on AFW pressure switches to TDAFW pump.

1512 Stopped TDAFW pump.

1513 Operators closed ERCW valves to TDAFW pump.

1516 Hotwell hi/lo level alarm cleared.

1517-1522 Low level alarms cleared on SGs.

1521 Closed main steam isolation valves (MSIV) due to continued cool down.

1714 MSIVs opened.

1729 Started standby Main Feedwater Pump (MFP).

1731 Stopped MDAFW pumps 1A-A and 1B-B.

D. Other Systems or Secondary Functions Affected

During this event, a minor issue involving the condensate booster pump (CBP) 1A was identified and was subsequently resolved. No other issues related to this event were identified.

E. Method of Discovery

Annunciation and indication in the Main Control Room.

F. Operator Actions

1. Operators manually tripped the turbine and entered AOI-17, "Turbine Trip."

2. Operators manually initiated AFW to restore SG levels.

3. Shortly after tripping the turbine, Operators manually tripped the reactor and entered E-0, Reactor Trip/Safety Injection, and transitioned to ES-0.1, Reactor Trip Response.

G. Automatic and Manual Safety System Response

When the unit operator observed that the turbine driven auxiliary feedwater pump had started and was aligned to the essential raw cooling water, the ASOS directed him to first verify adequate heat sink capability, and then stop the TDAFW pump. The UO then closed the ERCW valves to the TDAFW pump.

III. CAUSE OF EVENT

A. Immediate Cause

The cause of this event was the failure of the hotwell level control system to maintain an adequate water supply for the hotwell pump's net positive suction head due to a design deficiency.

B. Root Cause

1. Loss of Condensate Hotwell Pump Discharge Pressure

The root cause of this event is the hotwell level indicating high due to hydrodynamic effects of condensate, steam, and sparger spray flow on the hotwell level instrument taps inside the condenser (EIIS Code COND). This high indicated level caused the level control valve to open in automatic, flowing water from the hotwell to the condensate storage tank. This caused the level in the hotwell to lower to a point where the hotwell pumps lost net positive suction head.

After the turbine generator was synchronized to the grid at 1416, indicated hotwell level became erratic in the high direction. This caused level control valve 1-LCV-2-3 to open and dump back to the condensate storage tank (CST) from the hotwell pump discharge transferring hotwell inventory to the CST. At approximately 1456, indications

were observed of degraded hotwell pump performance. At 1457, hotwell pump discharge pressure dropped from 270 psig to 160 psig and condensate booster pump (CBP) 1A tripped.

The cause of the hotwell level indication failure was a deficient design of the sense line attachment location to the condenser which provided a false indication of high hotwell level

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as the main turbine is loaded. Additionally, the hotwell level control valve responded to the high level indication by opening to transfer water from the hotwell to the CST until hotwell level was too low to maintain adequate flow to the hotwell pump suctions. Operators did not transfer control to manual on the hotwell level control valve since the indicated level was high.

2. AFW Suction Pressure Switch Actuation

The root cause of the TDAFW pump pressure switches actuation are attributed to the MDAFW pump pulsations which are experienced at low flow conditions. These pulsations are manifested in the form of an acoustic wave which is amplified by the piping configuration to the TDAFW pump. Due to the rapid rate of pulsations, the pressure switches cannot react quickly enough to respond to the high pressure portion of the acoustic wave. Therefore, the pressure switches saw the wave form as a low pressure. This condition caused the switches 1-PS-3-121A and 1-PS-3-121D to actuate resulting in valves 1-FCV-3-136A and 1-FCV-3-136B being aligned to ERCW source.

IV. ANALYSIS OF EVENT - ASSESSMENT OF SAFETY CONSEQUENCES

A. Evaluation of Plant Systems/Components

Condenser Hotwell Level Control System

The hotwell level control system consists of three level transmitters and necessary piping and valves to allow the transfer of water between the hotwell and the CST. Level control valve, 1-LCV-2-9, which provides main condenser auto

makeup level control receives a signal from transmitter 1-LT-2-9 (EHS Code LT). This signal opens the valve to add water to the hotwell from the CST on decreasing hotwell level below setpoint.

Main condenser auto bypass level control valve, 1-LCV-2-3 (EHS Code LCV), receives a signal from transmitter 1-LT-2-3 (EHS Code LT) to open upon an increasing level in the hotwell above a setpoint allowing water to be transferred to the CST from the discharge of the hotwell pumps. When operating in automatic these two level control valves work together to maintain hotwell level between 27.1 and 28.9 inches which equates to approximately 56,000 gallons hotwell water inventory.

Level transmitter, 1-LT-2-12 (EHS Code LT), provides a signal to hotwell level recorder, 1-LR-2-12 (EHS Code LR), located on Panel 1-M-3 (EHS MCB) in the MCR. High and low level alarms annunciate in the MCR from signals generated by 1-LT-2-3 and 1-LT-2-9, respectively.

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The above three level transmitters and a local site glass utilize the same taps into the condenser. The lower tap penetrates the floor of the condenser approximately 8 inches from the south wall and approximately 8 inches east of centerline. The upper tap penetrates the south condenser wall approximately 6 feet above the lower tap. The upper tap is located in condenser zone C approximately 2 1/2 feet above the middle set of steam turning vanes. The hot well communicates with this area via the space between the vanes and small holes in the plate below the tube bundles (plate forms the roof of the hotwell).

During the event described by this licensee event report and a similar event which occurred on February 17, 1996 which resulted in only a turbine trip, the hotwell level control system failed to maintain level as designed which caused the loss of condensate flow due to voiding at the hotwell pump suction. In both events, indicated level in the hotwell began to trend high when the turbine generator was synchronized to the grid and increased as generator load was increased. This false high indication sent a signal to 1-LT-2-3 to open, diverting secondary plant water inventory back to the CST.

Approximately 31,000 gallons was transferred to the CST from

the hotwell over a period of about 30 minutes during this event. This amount of water equates to a decrease of approximately 16 inches of hotwell level. This combined with the fact that hotwell pump flows were increasing in response to loading the generator caused the loss of hotwell pump suction.

This erratic level control behavior is due primarily to the location of the sense line taps for the level transmitters. When the turbine is placed on line, steam and water flows increase dramatically in zone C of the condenser. In addition, No. 3 heater drain tank high level bypass sparger is located within a few feet of the upper level tap. The bottom sense line is located in a high flow area between the east and west sides of the zone C hotwell. These effects in combination appear to cause the level control problems. Even if the hotwell dumpback level controller was operated in manual, the high hotwell indication in the MCR would cause the Operator to manually dump hotwell inventory back to the CST resulting in the same problems described above.

Turbine Driven Auxiliary Feedwater Pump (TDAFW)

Water analysis of the AFW system, CST, and SGs indicated no introduction of ERCW into the AFW system. ERCW discharge pressure at the suction of the TDAFW pump was determined to be 8.8 psig. Based on the CST water level and suction piping configuration, it was determined that the TDAFW pump suction pressure (from the CST) was greater than the ERCW pressure. This condition coupled with the rapid operator response in shutting down the TDAFW pump prevented any detectable amounts of ERCW from entering the AFW system.

The MDAFW and TDAFW pumps discharge flows combine and flow through one common flow element. It was observed that the shut down of the TDAFW pump, in response to the opening of

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the ERCW isolation valves, resulted in a significant decrease in flow through the common flow element. This confirms that the MDAFW pumps were operating at low flow conditions. Recent testing demonstrated that at low forward flow from the MDAFW pumps, various suction supply test swap over pressure switches in the TDAFW pump suction did cycle in and out. During this testing, the swapover logic was not met.

It was concluded that suction pulsations occurred due to operating low MDAFW pump flow. These pulsations were of a magnitude great enough to actuate the controlling pressure switches for 1-FCV-3-136A and 1-FCV-3-136B, resulting in these valves going open.

B. Evaluation of Personnel Performance

The operating crew recognized the condensate/feedwater pressure reduction and took action to increase pressure to allow SG levels to be maintained by starting 1C hotwell pump immediately and reducing turbine load. Unable to maintain SG levels, the operators tripped the main turbine and began reducing reactor power with the capability of AFW. The crew then recognized that SG levels did not respond adequately and initiated a manual reactor trip.

Emergency Instruction E-0, "Reactor Trip/Safety Injection," was implemented and performed to the time of transition to ES-0.1, "Reactor Trip Response." While performing ES-0.1, the crew identified that ERCW had spuriously aligned to the suction of the TDAFW pump. After verification of adequate heat sink capability, the crew removed the TDAFW pump from service. Abnormal Operating Instruction (AOI) 17, "Turbine Trip," was also implemented and AOI-16, "Loss of Main Feedwater," was reviewed for necessary actions. Actions taken were appropriate and consistent with the plant procedures.

C. Safety Significance

This event was initiated by loss of hotwell pump suction eventually leading to loss of main feedwater. Operator proactive actions were manual turbine trip, power reduction, manual start of the MDAFW pumps, and manual reactor trip. Subsequent operator actions were manual control of AFW and MSIV closure due to reactor coolant system (RCS) (EIS Code AB) cool down.

Without operator action, the Reactor Protection System (EIS Code JE) would have terminated reactor power generation by the SG lo-lo level trip. Also, without operator action, AFW would have been automatically started to supply feedwater to the SG to maintain heat sink for the RCS. This transient also caused a cool down in which T sub ave went to approximately 529 degrees F. Evaluation of shutdown margin showed that, at beginning of life (BOL), the RCS can be cooled to 120 degrees F

before shutdown margin is lost and, at end of life (EOL) conditions, the RCS can be cooled to 463 degrees F before shutdown margin is lost.

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Operations followed ES-0.1 and boration in accordance with AOI-34, "Immediate Boration," to address the cool down below 537 degrees F. Had this event occurred at 100 percent reactor power, the plant would have been in an analyzed condition with more significant amounts of decay heat which could have diminished the rate of RCS cool down that occurred because of SG overcooling by the AFW system.

Procedures and operator actions were appropriate in this event from a safety significance position.

V. CORRECTIVE ACTIONS

A. Immediate Corrective Actions

The immediate corrective actions for this event were discussed in Section II.F, Operator Actions.

B. Corrective Actions to Prevent Recurrence

To address the hotwell pump loss of flow issue, TVA has taken the following corrective actions involving hardware modifications and procedure enhancements:

Hardware Modifications

TVA has implemented modifications to the existing level indication instrument taps to reduce the dynamic effects during hotwell transient conditions following initial generator loading. TVA has relocated the level standpipe and instrumentation from Unit 2 and installed this equipment on Unit 1, Zone A, for independent level indication which connected the upper tap within the hotwell area. In addition, TVA has removed the inner baskets from hotwell pump 1A and 1B suction strainers to minimize fouling.

Procedure Enhancements

TVA has revised the operating procedure to allow placing makeup and level control valves to CST closed and in manual prior to

turbine generator loading, and increased the hotwell level setpoint to accommodate potential inventory loss. In addition, TVA has revised the operating procedure to allow placing feedwater pump turbine mini-flow recirculation controller in manual to minimize impact on condensate and feedwater pressure and flow, prior to or shortly after, turbine generator loading, until plant conditions stabilize. TVA has also revised operating procedures to provide additional guidance for operators for onset of hotwell pump starving at less than 12 inches hotwell level at rated flow of 20,100 GPM.

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To address the TDAFW suction pressure switch action issue, TVA has taken the following corrective actions:

TVA has implemented a modification to use the MDAFW pump pressure switches for the TDAFW pump ERCW valves. TVA has received NRC approval of the Technical Specification change which occurred due to this modification. In addition, TVA has implemented a spring check valve (EISS Code V) modification in AFW pump recirculation lines and repacked 1A and 1B MDAFW pumps.

VI. ADDITIONAL INFORMATION

A. Failed Components

1. Safety Train Inoperability

There were no failures that rendered a train or a safety system inoperable.

2. Component/System Failure Information

a. Method of Discovery of Each Component or System Failure:

Hotwell Indication Failure - The root cause for this failure was determined by thorough analysis of the event data and through inspection of the equipment. The investigation team used Kepner-Tregoe (KT) analysis techniques to determine the root cause of the event.

b. Failure Mode, Mechanism, and Effect of Each Failed

Component:

The failure of the hotwell indication was due to a design deficiency. The placement of the instrument taps for this indication did not consider the previously described hydrodynamic effects during the initial loading of the turbine and the resulted effect on the hotwell level control system.

c. Root Cause of Failure:

Design error - The placement of the instrument taps caused inaccurate readings to be provided to both the main control room and the hotwell level control system.

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d. For Failed Components With Multiple Functions, List of Systems or Secondary Functions Affected:

1. The hotwell instrument taps location caused the level transmitter 1-LT-2-9 to provide a false signal to level control valve 1-LCV-2-9 which opens the valve to allow flow from the condensate storage tank to the hotwell.

2. The hotwell instrument taps location also caused the level transmitter 1-LT-2-3 to provide a false signal to level control valve 1-LCV-2-3 which opens the valve to allow flow from the hotwell to the condensate storage tank.

2. The hotwell instrument taps location resulted in level transmitter 1-LT-2-12 providing a false signal to hotwell level recorder 1-LR-2-12 located in the main control room.

e. Manufacturer and Model Number of Each Failed Component:

Rosemount, Inc., Transmitter Model No. 1151DP4B22PB
General Electric Company, Recorder Model No. 530

B. Previous Similar Events

For Watts Bar Unit 1, no events similar to the events described in this report have been previously reported under 10 CFR 50.72 or 10 CFR 50.73.

Although not reportable under 10 CFR 50.73, a similar event occurred on February 17, 1996, while the plant was in Mode 1 at normal operating temperature and pressure of 557 degrees F and 2235 psig, respectively. This event involved similar erratic hotwell level indication which ultimately resulted in a manual turbine trip.

The difference in that event and the one described by this LER, is that it occurred over a longer period of time. This meant that the operators were able to control hotwell level more effectively by closing the dump back valve to the condensate storage tank. At that time, the root cause for that event was considered to be foreign material found in the hotwell pump suction strainers which caused a loss of flow. After subsequent review, that event was most likely the result of the same root cause as was the event described by this LER.

VII. COMMITMENTS

The actions taken in response to this event are tabulated in Section V, Corrective Actions. These actions are complete.

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TVA
Tennessee Valley Authority, Post Office Box 2000,
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MAR 21 1996

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
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Gentlemen:

In the Matter of) Docket Nos. 50-390
Tennessee Valley Authority)

WATTS BAR NUCLEAR PLANT (WBN) - UNIT 1 FACILITY OPERATING
LICENSE NPF-90
- LICENSEE EVENT REPORT (LER) 50-390/96004 - MANUAL REACTOR TRIP

AND
RELATED ENGINEERED SAFETY FEATURE (ESF) ACTUATIONS

The purpose of this letter is to provide the subject LER. The enclosed report provides details regarding the manual reactor trip and related ESF actuations which occurred on February 19, 1996, after the initiation of a manual reactor trip. Submittal of this report is in accordance with 10 CFR 50.73(a)(2)(iv).

If you should have any questions, please contact P. L. Pace at (423) 365-1824.

Sincerely,

D. V. Kehoe
Nuclear Assurance
and Licensing Manager

Enclosure
cc: See page 2

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U.S. Nuclear Regulatory Commission
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ENCLOSURE

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